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September 29, 2008

L-PI-08-076
10 CFR 50.73

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Prairie Island Nuclear Generating Plant Unit 1
Docket 50-282
License No. DPR-42

LER 1-08-02, Inadvertent Reactor Trip Caused by Failed Controller During Reactor Protection System Testing

Licensee Event Report (LER) 1-08-02 for this event is attached. Northern States Power Company, a Minnesota corporation (NSPM), notified the NRC of this event as required by 10 CFR 50.72(b)(2)(iv)(B) on July 31st, 2008. Please contact us if you require additional information related to this event.

Summary of Commitments

This letter contains no new commitments and no changes to existing commitments.

Michael D. Wadley

Michael D. Wadley
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
Department of Commerce, State of Minnesota

ENCLOSURE

LICENSEE EVENT REPORT 1-08-02

4 Pages Follow

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES: 08/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0066), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Prairie Island Nuclear Generating Plant, Unit 1	05000282	1 of 4

4. TITLE
Inadvertent Reactor Trip Caused by Failed Controller During Reactor Protection System Testing

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
07	31	2008	2008 –	002 –	00	09	29	2008	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: Mode 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☑ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Jorge L. O'Farrill, Licensing Engineer	651.388.1121

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	JC	IMOD	F180	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
○ YES (If yes, complete 15. EXPECTED SUBMISSION DATE).	⊙ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On July 31, 2008, Prairie Island Nuclear Generating Plant (PINGP), Unit 1 was operating at 100 percent power. At 0817 CDT during performance of the quarterly analog protection functional test, Unit 1 reactor tripped. At the time of the trip the yellow channel over-temperature delta T (OT delta T) bistables were in the tripped condition as directed by the testing procedure when a red channel OT delta T reactor trip signal was generated due to a failed controller. The red channel OT delta T setpoint was not expected to be challenged nor was a reactor trip expected at any point during yellow channel OT delta T analog testing.

All automatic actions for a reactor trip occurred as required with two exceptions: The Unit 1 turbine-driven auxiliary feedwater pump (11 TDAFWP) auto started as designed, but tripped 42 seconds later on low discharge pressure (see LER 1-08-03); and a Unit 1 Turbine 2 Reheat Stop Valve indicated intermediate vice closed due to a fault with the position indication. Operator response and recovery actions were as expected.

The reactor trip was caused by a failed F delta Q proportional controller, which was subsequently replaced. Planned corrective actions include replacement of all similar controllers.

EVENT DESCRIPTION

On July 31, 2008, 0817 CDT, Prairie Island Nuclear Generating Plant (PINGP), Unit 1 was operating at 100 percent power when the reactor tripped on an over-temperature delta T (OT delta T) reactor trip signal from the reactor protection system[1].

At the time of the event, instrumentation and control personnel were performing the quarterly analog reactor protection functional test on the yellow channel when the red channel OT delta T bistable was actuated. Subsequent troubleshooting and root cause investigation determined that the red channel bistable actuation was caused by the failure of a Foxboro H-line (model 62H-2E-O) F delta Q controller[2] in the OT delta T circuit. The controller output failed high causing the OT delta T setpoint to drop below the actual delta T parameter thus causing a red channel reactor trip signal. The red channel OT delta T reactor trip signal combined with the yellow channel OT delta T bistables being in test (trip) as directed by the surveillance procedure completed the 2 out of 4 coincidence logic required to initiate a reactor trip. During the performance of yellow channel OT delta T analog testing, the red channel OT delta T setpoint was not expected to be challenged nor was a reactor trip expected at any point.

All automatic actions for a reactor trip occurred as required with the following exceptions: Subsequent to the trip, the Unit 1 turbine-driven auxiliary feedwater pump (11 TDAFWP) auto started as designed, but tripped 42 seconds later on low discharge pressure. And a Unit 1 Turbine 2 Reheat Stop Valve indicated intermediate vice closed. However, physical inspection verified that this valve was indeed closed and that the intermediate indication was caused due to a failed switch rod (linkage) that actuates a proximity switch to indicate valve position. Operator response and recovery actions for the reactor trip were completed as expected.

EVENT ANALYSIS

A reactor trip is required to be reported per 10 CFR 50.73(a)(2)(iv)(A). The reactor trip by itself did not result in a condition that could have prevented the fulfillment of a safety function per 10 CFR 50.73(a)(2)(v). Issues associated with the 11 TDAFWP are addressed in LER 1-08-03. The erroneous indication for the Unit 1 Turbine 2 Reheat Stop Valve is not directly related to the reactor trip and was repaired under the site's corrective action program on 08/02/2008.

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[1] EIIS System Code: JC
[2] EIIS Component Identifier: IMOD

SAFETY SIGNIFICANCE

The OT delta T trip along with the overpower delta T trip is designed to keep the departure from nuclear boiling ratio (DNBR) greater than the limit for slow reactivity additions. This event was due to an equipment failure and not related to a reactivity addition. With the exception of the 11 TDAFWP trip and a Unit 1 Turbine 2 Reheat Stop Valve position indication, all systems performed as expected to the reactor trip signal and operators responded and recovered as expected. Thus, this event did not affect the health and safety of the public and the safety significance of this event is considered minimal.

CAUSE

The equipment root cause for the failure of the F delta Q controller is attributed to the random failure of varactor diode (CR1) located inside the controller. Although this controller was refurbished in 1985, only the capacitors were routinely replaced as part of refurbishments. Therefore, CR1 was not replaced as part of the 1985 refurbishment.

The organizational cause was found to be the inadequate prioritization by the site in the creation of a preventive maintenance strategy for the analog components within the reactor protection and control system.

CORRECTIVE ACTION

Immediate corrective action:

1. Replaced the failed F delta Q proportional controller.

Planned corrective actions include:

1. Replacement or refurbishment of all F delta Q proportional controllers.
2. Implement an improved preventive maintenance strategy for the Foxboro H-Line components of the reactor protection and control system.
3. Implement a Life Cycle Management Plan for the reactor protection and control system. This will ensure timely preventative replacement of the Foxboro H-Line components.

PREVIOUS SIMILAR EVENTS

LER 2-07-01 describes a reactor trip due to the failure of an MG-6 style relay in the safety injection system[3] and LER 1-06-01 describes a reactor trip due to a ground caused by degraded motor insulation in one of the condensate system[4] pumps.

Although both of these events describe reactor trips due to equipment related issues, the MG-6 style relay failure was due to high contact resistance while the ground caused by degraded motor insulation was an age related failure. Both of the previous LERs include preventive maintenance in the corrective actions. LER 1-06-01 included an action to institute a large motor program that would not prevent the event of this LER. LER 2-07-01 included an action to implement preventive maintenance strategy for all critical equipment. This action was completed in February of 2008, but some improvements were made apparent by the event of this LER (see Corrective Action discussion, above).

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[3] EIIS System Code: BQ
[4] EIIS System Code: SD